ACCESSION #: 9901270180

NON-PUBLIC?: N

LICENSEE EVENT REPORT (LER)

FACILITY NAME: COMANCHE PEAK STEAM ELECTRIC STATION PAGE: 1 OF 5

UNIT 2

DOCKET NUMBER: 05000446

TITLE: MANUAL REACTOR TRIP DUE TO TURBINE LOAD SWINGS

EVENT DATE: 01/03/99 LER #: 99-002-00 REPORT DATE: 01/20/99

OTHER FACILITIES INVOLVED: CPSES UNIT 1 DOCKET NO: 05000445

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: Raphel Flores, System Engineering TELEPHONE: (254) 897-5590

Manager

COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:

REPORTABLE TO NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At approximately 11:14 a.m., on January 3, 1999, Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 1, Power Operation, with reactor power at approximately 100 percent, main turbine generator control valves suddenly and unexpectedly closed while at 100 percent rated power. As designed, all steam dumps immediately opened and

rods automatically stepped in to control the reactor coolant system temperature transient. The reactor plant, designed to withstand a 50 percent load rejection without a trip, experienced an equivalent 80 percent load rejection event. The cause of the load rejection was not apparent. In response to the loss of turbine load control, the Unit 2 Operations Shift Supervisor (utility, licensed) made a conservative decision to manually trip the reactor.

The cause of the event could not be determined. A data acquisition system (minidas) has been connected to the Electro-Hydraulic Control (EHC) control system to monitor and collect data and provide additional diagnostic information, should the failure repeat.

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

Any event or condition that resulted in manual or automatic

actuation of any Engineered Safety Feature (ESF), including the

Reactor Protection System (RPS)(EIIS:(JC)).

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On January 3, 1999, Comanche Peak Steam Electric Station (CPSES)

Unit 2 was in Mode 1, Power Operation, with reactor power at

approximately 100 percent.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE

AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

There were no inoperable structures, systems, or components that

contributed to the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE

TIMES

At approximately 11:14 a.m. (CST) on January 3, 1999, the

Comanche Peak Steam Electric Station (CPSES), Unit 2, main turbine generator control valves suddenly and unexpectedly closed while at 100 percent rated power. As designed, all steam dumps immediately opened and rods automatically stepped in to control the reactor coolant system temperature transient. The reactor plant, designed to withstand a 50 percent load rejection without a trip, experienced an equivalent 80 percent load rejection event, however, the turbine runback alarms did not annunciate. Shortly after the control valves had closed (approximately 5 seconds), the main turbine control system began operating as if it had been transferred to the speed control mode by controlling main generator output to 200 MWe. The main turbine generator speed control mode resulted in the control valves cycling between the fully closed position to 3 percent open every few seconds. About 9 seconds into the event, one pressurizer power-operated relief valve opened for approximately 5 seconds to control the resultant reactor coolant system pressure increase. As pressurizer level oscillated from the cyclic operation of the control valves, reactor coolant system pressure continued to slowly decrease. After one minute into the event, the Unit 2 Operations shift supervisor (utility, licensed) ordered the manual trip of Unit 2. The auxiliary feedwater system automatically started when steam generator levels dropped following the manual trip. All systems responded normally. The

unit was stabilized in Mode 3, Hot Standby.

An event or condition that results in an automatic actuation of

an ESF including the RPS, is reportable pursuant to

10CFR50.72(b)(2)(ii). On January 3, 1999 at approximately 1:54

p.m. CST, the Nuclear Regulatory Commission Operations Center

was notified of the event via the Emergency Notification System.

E. THE METHOD OF DISCOVERY OF EACH COMPONENT FAILURE, OR PROCEDURAL

OR PERSONNEL ERROR

The Steam Generator steam flow/feed flow mismatch alarms alerted

the Balance of Plant operator(utility, licensed) of the event.

II. COMPONENT OR SYSTEM FAILURES

A. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT

The cause of the Electro-Hydraulic Control (EHC) failure which

caused the event could not be determined. Additionally, the

cause of the failure of the turbine runback alarm to annunciate

could not be determined.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

The cause of the Electro-Hydraulic Control (EHC) failure which

caused the event could not be determined. Additionally, the

cause of the failure of the turbine runback alarm to annunciate could not be determined.

C. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF

COMPONENTS WITH MULTIPLE FUNCTIONS

Not applicable - The cause of the Electro-Hydraulic Control

(EHC) failure which caused the event could not be determined.

Additionally, the cause of the failure of the turbine runback

alarm to annunciate could not be determined.

D. FAILED COMPONENT INFORMATION

Not applicable - A failure of the component could not be

ascertained.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

The following safety system actuations occurred as expected as

a result of this event. Auxiliary Feedwater System

(AFW)(EIIS:BA).

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

Not applicable - there were no systems or components that were

inoperable that contributed to this event.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

This event is well within the limits of the analyses described

in Section 15.2.2 of the CPSES Final Safety Analysis Report

(FSAR) for "Loss of External Electrical Loads." The Nuclear Steam Supply System can safely withstand a full load rejection. The event of January 3, 1999, occurred at 100 percent power, and all systems and components functioned as designed. Based on this analysis it was concluded that this event did not adversely affect the safe operation of CPSES Unit 2 or the health and safety of the public.

IV. CAUSE OF THE EVENT

Despite extensive troubleshooting performed by the EHC vendor and the CPSES maintenance department, the cause of the unexpected turbine load change for this trip and the similar trip (reference CPSES LER 50-446/96-003-00) which occurred on February 23, 1996 remains indeterminate.

The PORV that opened was the one fed with a lead/lag compensated pressure error signal. This PORV is more likely to open in such transients where there is a rapid rate of pressure increase, even though the actual pressure did not exceed 2335 psig. The cause of the pressurizer power-operated relief valve opening was postulated to the pressure transient above normal pressure. The pressurizer power-operated relief valve seated properly with no indication of seat leakage.

V. CORRECTIVE ACTIONS

The immediate corrective actions were to manually trip the plant and stabilize it in Mode 3. The EHC system has been connected to the minidas (a data acquisition equipment) to monitor

equipment performance, for some period of time. If a similar load reduction occurs, the captured data will be utilized to assist in the evaluation of causes. Additionally, approximately 5 controller cards were removed and replaced to assess the cause, no matters of concerns with respect to these cards could be determined.

VI. PREVIOUS SIMILAR EVENTS

The February 23, 1996 event (reference CPSES LER 50-446/96-003-00) at CPSES is similar to this event. Both events involve Turbine/ Generator load swings resulting in a manual reactor trip. Troubleshooting efforts associated with the February 23, 1996 event led to the postulation that faulty printed circuit cards in the Electro-Hydraulic Control (EHC) runback circuits may have caused the event, however, results were inconclusive. Therefore, it was ascertained even though the events were similar the cause of the events are believed to be different.

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Log # TXX-99014

TUELECTRIC File # 10200

Ref. # 10CFR50.73(a)(2)(iv)

C. Lance Terry

Senior Vice President

& Principal Nuclear Officer January 20, 1999

U. S. Nuclear Regulatory Commission

Attn: Document Control Desk

Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)-UNIT 2

DOCKET NOS. 50-446

MANUAL OR AUTOMATIC ACTUATION OF ENGINEERED SAFETY FEATURES

LICENSEE EVENT REPORT 446/99-002-00

Gentlemen:

Enclosed is Licensee Event Report (LER) 99-002-00 for Comanche Peak Steam

Electric Station Unit 2, "Manual Reactor Trip due to Turbine Load

Swings."

This communication contains no new licensing basis commitments regarding

CPSES Unit 2.

Sincerely,

C. L. Terry

By:

Roger D. Walker

Regulatory Affairs Manager

OB:ob

Enclosure

cc: Mr. E. W. Merschoff, Region IV

Mr. J. I. Tapia, Region IV

Resident Inspectors, CPSES

COMANCHE PEAK STEAM ELECTRIC STATION

P.O. Box 1002 Glen Rose, Texas 76043-1002

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